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FAST BREEDER BLANKET FACILITY
FBBF

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MASTER

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ABSTRACT

The work performed in the reporting period was primarily concerned with the preparation of the experiments to be performed on the Fast Breeder Blanket Facility (FBBF) and the corresponding analysis. After some initial delays, the work appears now to be on schedule.

The work on the experimental program has been started. Since experiments are subject to safety constraints, a safety investigation program (for a hypothetically flooded facility) has to be established and tested. This work is now completed and is reported (A). The neutronics part of the preanalysis is also reported (B).

The testing of the first configuration has largely been prepared (C). The identification of the experiment need has been worked on extensively, largely through unsponsored research which had been started before the contract became effective (D). The work done in this area by other groups is currently being reviewed (F).

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Task A (K. O. Ott, R. C. Borg, F. M. Clikeman, R. H. Johnson):

Task A consists of outlining the "detailed experimental program," as far as it can be defined without "detailed pre-analysis;" the first-year work-scope included pre-analysis only for the first loading. Clearly, the pre-analysis is to be used to optimize the lay-out of each specific loading as well as the combination of various loadings, so that the most useful results can be derived from the FBBF project. Therefore, the experimental program of Task A is detailed in the overall structure of its lay-out. But, the specific features of the various configurations as well as the most desirable sequencing of the different loadings is to be found by pre-analysis of a larger phase of the program.

The overall experimental program includes the two basic categories of integral experiments:

- (1) "Clean" experiments, which are laid out for unquestionable comparability with respective theoretical calculations.
- (2) "Engineering mock-up" type of experiments, which are laid out to closely simulate actual complicated configurations as they may occur e.g. in rotating subassemblies with skewed plutonium distribution.

Emphasis is first placed on the clean integral experiments, which have the character of "benchmark experiments."

The detailed experimental program is being developed under two guidelines:

- (1) The experiments must be safe in the sense that k_{eff}^F , i.e. the k_{eff} of a hypothetical complete flooding of the FBBF, remain well below 1, including computational errors.

- (2) The differences between the various FBBF configurations should be determined such that they provide experimental information which is suitable for theoretical interpretation.

The "detailed experimental program" is currently being developed and will be reported around the end of September. An important role in the lay-out of this program is played by the safety investigation to assure sufficient subcriticality. The method applied for these investigations is briefly reported in the following.

A.1 Method for k_{eff} Calculation of a Flooded and Non-flooded Assembly for Task A (R. C. Borg, O. H. Gailar, and J. E. Arpa)

Safety is the chief factor governing whether or not an experimental loading may be performed. Therefore presafety evaluation of each loading must be carried out before it is set up. This section indicates some of the calculational procedures used for the safety calculations for the initial loading.

Fig. A.1 shows a horizontal cross-section for the initial loading of FBBF. The two spectrum transformer annuli contain enriched 4.8% fuel while the blanket contains fuel at natural enrichment. The spaces between fuel rods in the inner transformer annulus are almost completely filled with sealed sodium-containing stainless steel tubes. The two transformer annuli are separately canned and sealed. Because of this type of construction, the possibility of flooding any region is very remote. As a further safety precaution, the floodable volume between the fuel rods in the blanket containers has been reduced to a minimum by inserting SS tubes around each fuel rod.

For this study it was postulated that any or all of the annuli could be accidentally flooded. Flooding of all annuli is assumed in the available space around the fuel pins and the stainless steel tubes.

- 1 SOURCE REGION
- 2 INNER TRANSFORMER REGION
- 3 OUTER TRANSFORMER REGION
- 4 INNER BLANKET
- 5 OUTER BLANKET
- 6 REFLECTOR

Fig. A.1 Horizontal cross-section of FBBF

The fuel pins are arranged in a very tight lattice and, under dry (not flooded) conditions, the neutron energy spectrum will be characteristic of that of a fast reactor. In the event of flooding, the spectrum will be softer than in the nonflooded case. However, because of both the tight lattice and the presence of sealed sodium-containing tubes between fuel pins in one region of the facility, and the additional SS-cladding in the blanket, the available volume for flooding water is severely limited, and the spectrum will not be as soft as it is in a thermal reactor.

The computational method must be applicable to a variety of conditions. All regions of the facility are separated by solid partitions, it is therefore conceivable that during or after a flooding accident some regions may be flooded and others may not be flooded. One expects that the highest reactivity configuration is achieved if all regions are flooded, however, to demonstrate this, various flooded-nonflooded combinations must be considered. Since the configuration is strongly undermoderated for flooded conditions the computational method must be applicable to both an intermediate and fast spectrum.

1. The Computational Method

The computational method includes the following:

1. Definition of a unit cell for each type of fuel region.
2. Calculation of group cross sections characteristic of these unit cells, for both flooded and nonflooded conditions.
3. Few-group, one-dimensional diffusion calculations with all fuel regions and reflectors represented. Repetition for various flooded-nonflooded combinations.
4. Selection of extreme cases (maximum k_{eff}) from above and two-dimensional (r-z) diffusion calculations representing trans-former, blanket, and reflector regions explicitly.

It is expected that the completely flooded case is the most reactive (greatest k_{eff}). A computational method, applicable to this condition (i.e., a flooded but low water-to-metal ratio cell), the HAMMER¹ program, was chosen. It has two chief advantages:

1. an integral transport treatment of the heterogeneous cell for both thermal and fast neutrons;
2. a combination of fast and thermal calculations in one computer run.

HAMMER uses the MUFT² treatment* and has historically been applied to thermal reactors but not to those with zero or low hydrogen content. The validity of using HAMMER for the FBBF calculation was investigated by comparing results of similar HAMMER and HRG³ calculations, where HRG is more applicable to dry cores. Table A.1 illustrates the results and shows HAMMER and HRG to be in very good agreement.

The validity of the HAMMER approach for a fast spectrum is further supported by the comparison with the results reported in Task B, which have been calculated with standard fast reactor computational methods (comp. Task B, Sec. 4). As an integral test of HAMMER, the driest of several YANKEE experiments was analyzed using the computational method described here. Table A.2 illustrates that these results were in good agreement with the experiment.

2. Verification of the Method

The YANKEE critical with a water-to- UO_2 volume ratio of 1.036:1 was chosen as the low-water content benchmark problem. The unit cell was represented by a 0.381 cm radius UO_2 fuel pin in a 0.00483 cm thick stainless clad in a cell with a radius of 0.580 cm. The hydrogen-to-²³⁵U ratio was 115.31:1. The HAMMER-calculated macroscopic cross sections

* exact treatment for hydrogen, Greuling-Coertzel for other scatterers.

Table A.1 Comparison of HAMMER (MUFT) AND HRG
for Various Water Densities

Case	k_{∞} (HAMMER) (Fuel Cell Only)	N_H ($\times 10^{24}$)	N_H/N_{235}	k_{eff} (HAMMER- group constants)	k_{eff} (HRG-THERMOS group constants)
1. Dry Cell, Al replaces Na	0.678	-	-	0.4187	0.4148
2. Flooded	0.775	.00285	7.70	0.5148	0.5215

No. of Groups	Water (reflector) Cross Sections	Calculated Critical Radius (cm.)	Calculated k_{eff} at experimental critical radius	Reactivity difference between calculation and experiment
4	From H ₂ O region in HAMMER	32.61	0.9965	-.0038
4	From HAMMER for H ₂ O alone	33.77	0.9893	-.0107
2	From HAMMER for H ₂ O alone	33.46	0.9916	-.0084

Table A.2 Critical size and effective multiplication factors calculated for YANKEE critical experiment.

were used in a two-region radial diffusion calculation performed by the FOG⁴ program. The fuel region was surrounded by a thick water reflector, and a critical dimension search on the core-reflector interface was carried out. The results of these calculations are given in Table A.2. Inspection of the table shows that depending upon the number of groups used and the origin of the cross sections, the calculated critical radius differs from the measured value by from about 2 to 6%, corresponding to differences in k_{eff} of from -0.0035 to -0.0107. This study demonstrates that the overall process to be used in the FBBF analysis, when applied to a known thermal reactor gives results that are in reasonably good agreement with measurements.

3. Substitution of Aluminum for Sodium

The HAMMER fast library does not contain sodium. For this reason aluminum was substituted for sodium. The number density of the Al was chosen to conserve the slowing down power of the Na it was replacing, i.e.,

$$(\xi \Sigma_s)_{Al} = (\xi \Sigma_s)_{Na}$$

Values of ξ and σ_s were obtained from the group constant set of Ref. 5, and the average value of $\xi \Sigma_s$ was conserved over the energy range of from 6.5 MeV. to 0.465 ev. It was found that the Al number density required for this conservation was 4.331 times that of the Na. The neutron capture in both Na and Al is negligible and has therefore no impact on the substitution.

4. Comparison of HAMMER(MUFT) with HRG

This work was done to investigate the validity of applying the MUFT slowing down treatment in a low or zero hydrogen environment. The

outer transformer region was used for the comparison. Figure A.2 illustrates the outer transformer cell and its equivalent HAMMER cell. Resulting group constants were used in 4-group, three-region radial FOG diffusion calculations. The FOG geometry consisted of a center water region, a fuel annulus, and an outer water region. The fuel annulus dimensions were those of the FBBF outer converter region.

Fast group constants were obtained from HRG or the fast edit of HAMMER (MUFT). Both calculations used the same thermal group constants from the HAMMER edit. Thus, the difference between the two calculations was due to fast group constant differences which, in turn, were caused by the different slowing-down treatments used in MUFT and HRG and by the different cross section libraries employed by the two programs. The results of these calculations are given in Table A.1.

The agreement between the k_{eff} values obtained from the MUFT and HRG treatments is very good, and strengthens the argument for using the more convenient HAMMER program.

5. Application to FBBF

The unit cell shown in Figures A.2 and similar equivalent cells for the inner transformer and the blanket regions were used as input to HAMMER. Both flooded and dry cases were calculated. Only physically floodable regions, for example, region 4 in Figure A.2, were flooded. The water density in all cases was unity. A series of two- and four-group multi-region, radial FOG diffusion calculations were performed for various combinations of flooded-dry regions. The regions, from center out, are indicated by the heading given in Table A.3. For the flooded blanket cases the reflector was 40 cm. thick and a zero flux was assumed at the outer edge of the blanket. The radii of the various regions were

Region	Material	Dimension(cm)	Volume Fraction
1	Fuel	0.5334	0.3246
2	Gap	0.5410	0.0093
3	Clad	0.5918	0.0657
4	Void (Water)	0.6226	0.0426
5	Stainless Steel	0.6826	0.0894
6	Na (Al)	0.9362	0.4624

Fig. A.2 Outer converter cell of FBRF

Table A.3 Effective Multiplication Factors for Various
Flooded-Dry Conditions
One Dimensional Calculations

Case No.	Source	Inner Core	Outer Core	Blanket	Fe Reflector	Reflector	k_{eff}
1	-	F	F	F	-	F	0.6770
2	-	F	F	D	-	D	0.6827
3	-	D	F	F	-	F	0.5082
4	-	D	F	D	-	D	0.4864
5	-	F	D	F	-	F	0.6693
6	-	F	D	D	-	D	0.6746
7	-	D	D	F	-	F	0.4884
8	-	D	D	D	-	D	0.4585
9	F	F	F	F	SS	F	0.7180
10	SS	F	F	F	SS	F	0.6486
11	SS	D	D	D	SS	D	0.4219

F - Flooded

D - Dry

- - Not Applicable

fixed at their respective design values. The effects of axial blankets and reflectors were estimated by input transverse bucklings.

The effective multiplication factors obtained from these calculations are presented in Table A.3. These multiplication factors, for all possible flooded-dry combinations, are all well below unity, indicating that the facility will remain well below critical when flooded in the worst (most reactive) case.

Since the one-dimensional FOG calculations could not explicitly represent the axial blanket and reflector, two-dimensional (r-z) calculations were performed for the flooded and non-flooded cases. These r-z calculations were carried out with four neutron energy groups using the EXTERMINATOR-II⁶ program. In both cases, and in their corresponding FOG runs, explicit regions representing the source and a radial iron reflector were added (see Fig. A.3). The resulting k_{eff} values, as indicated in Table A.4, were 0.416 and 0.710 for the dry and flooded cases respectively. These values are in excellent agreement with their one-dimensional counterparts.

6. Conclusion

The calculational methods have been verified and applied to specific cases for various flooding configurations in the FBBF. Effective multiplication factors below 0.75 were obtained for the most reactive situation studied indicating that even under the worst flooding accident conceivable, the initial FBBF loading will remain well below critical.

Fig. A.3 Geometry for r-z calculations using EXTERMINATOR-II.

Table A.4 Effective Multiplication Factors for Various
Flooded-Dry Conditions
Two-Dimensional Calculations

Case	Source	Inner Core	Outer Core	Blanket	Fe Reflector	Reflector	k_{eff}	k_{eff}^+
1	SS	D	D	D	SS	D	0.416	.393
2	F	F	F	F	SS	F	0.710	

$^+k_{eff}$, see Task B sec. 4

F - Flooded

D - Dry

List of References for Task A

1. J. E. Suich and H. C. Honeck, "The HAMMER System", DP-1064, January 1967.
2. MUFT is the Multi-Group Fourier Transform Program used in the fast section of HAMMER.
3. HRG is a Hanford revision of GAM-1, supplied by J. L. Carter. G. D. Joanou, E. J. Leshan, and J. S. Dudek, "GAM-1, a consistent multigroup code for the calculation of the fast neutron spectrum and multigroup constants", GA-1850, May 1961.
4. H. P. Flatt, "The FOG One-Dimensional Neutron Diffusion Equation Codes", NAA-SR-6104.
5. L. P. Abagyan, N. O. Bazazyants, I. I. Bondarenko and M. N. Nikolaev, "Group Constants for Nuclear Reactor Calculations", Edited by Professor I. I. Bondarenko, Consultants Bureau, New York, 1964.
6. T. B. Fowler, M. L. Tobias and D. R. Vondy, "Exterminator-II, A Fortran IV Code for Solving Multigroup Neutron Diffusion Equations in Two Dimensions, ORNL-4078, April 1967.

TASK B (R. C. Borg and D. J. Malloy):

The object of this task is to perform a detailed preanalysis of the first experimental sequence. The neutronics-calculation part of the preanalysis is reported here.

B.1 Neutron Flux Calculations for Initial FBBF Loading

The neutronics calculation for the initial FBBF loading has been performed using six energy groups, neutron diffusion theory, and a two-dimensional geometric model. Due to the fact that the FBBF has a cylindrical shape, a radial symmetry exists about its axial center line. Thus, a two-dimensional representation (i.e. r-z geometry) is sufficient for the purposes of a neutronics calculation.

A special six-group constant set has been devised to find the multiplication constant, k_{eff} and the leakage flux. The six-group cross sections have been obtained by collapsing a 21-group cross section set down to the six-group level by weighting the cross sections with a representative spatial flux for the region of interest. In order to obtain a reasonable representative spatial flux, a one-dimensional calculation was made with 21 energy groups. The 21 group cross sections were then collapsed to the 6 group level over this one-dimensional spatial flux solution.

The FBBF has several distinct regions which have unique physical properties (e.g. transformer regions, blanket regions, reflector, etc.). These regions were each represented by a homogenization of their individual materials. Thus, the heterogeneous system is represented in the calculational model by several unique homogenized materials (see Figs. B.1 and B.2). This homogenization procedure is quite valid for a system which exists in a fast neutron environment. The average neutron energy in the FBBF is relatively high, thus the average mean free path is large. Further, the fuel pins are packed together in a fairly tight lattice. The combined effect of

Figure B.1: Cut-away view of the PDBF.
The neutron source is shown
in the lowered position.

Figure B.2: FBBF two-dimensional mock-up (to scale)

these two conditions is a "homogenization" of the materials. Thus, the computational homogenization procedure closely reflects the physical situation and is quite accurate.

A detailed description of the design configuration used in the calculation is given below. Following this is a description of the one-dimensional and two-dimensional calculations. Also included is a short discussion on the cross section sets employed and the global results of the two-dimensional calculation.

1. Design Configuration

Figure B.2 illustrates the two-dimensional representation of the facility. The radial dimensions are to design accuracy. The axial dimensions include the following assumptions and/or simplifications which were necessary:

1. The concrete height of 20 cm is arbitrary. However, since the relaxation length (i.e. the decrease in the neutron flux by a factor $e = 2.71$) of concrete is approximately 10 cm, an increase in this dimension should have little practical effect on the calculation.
2. The source plug of steel is only represented up to the active blanket height of 4 ft. and not in its full axial length.
3. Similarly, the axial blanket and reflector are both represented to a height of 4 ft. above the transformer region/blanket support plate.
4. The transformer region/radial blanket support plate is modeled as a 3 in. thick steel plate.

The regional compositions (i.e. the actual design materials of the facility) are listed below.

1. Source Region/Source Steel - a homogenization of the source mock-up is depicted in Figure B.3. The volume fraction of steel is 1.0 in the source region and 0.3305 in the source steel region below the source. This latter value represents the source can wall only (i.e. the source pellets and container are removed).
2. Concrete - the nominal concrete composition is as follows:

element	elemental density	
	g/cm ³	
H	.02	
C	.118	
O	1.116	
Na	.011	
Fe	.026	
Al	.085	Total Density: 2.368 g/cm ³
Mg	.057	
Ca	.58	
Si	.34	
K	.004	
S	.007	

3. Structural Steel (1020) - Normal Carbon Steel: C: .2%; Mn: .45% Fe: 99.35%.
4. Inner Transformer Region - Tightly packed lattice (pitch = 0.4665"), 4.8% enriched, active length = 36", SS-348 clad.
5. Outer Transformer Region - 4.8% enriched pins; packed to a pitch of .702"; SS-348 clad; SS-316 interstitial cans, filled with sodium.

Figure B.3 Source Region mock-up (to scale)

6. Blankets (radial and axial) - Natural uranium rods, constant pitch of .672953" throughout blanket, active length = 48" for radial, 12" for axial. (Instead of the axial uranium blanket a less reactive axial lead blanket will be used for the first configuration.)
7. Reflector - 66% structural steel (1020), 34% salt (NaCl) per volume.
8. Source fillet at top of facility - steel.

2. Calculation Model

Certain elements which are in peripheral regions of the FBBF design are not available in the 21 group cross section set which is being utilized (ENDF/B, version IV data). Thus, these elements had to be represented with available elements. The materials in each region of the calculational model are described below.

1. Source Region/Source Steel: the source capsule is SS-304, the surrounding cans are SS-316. Nominal compositions of each steel were homogenized to produce the appropriate number densities. (See below for the non-homogenized individual nominal steel number densities). The volume fraction of SS-304 is 1.0 for $r < 0.635$ cm; the volume fraction of SS-316 is 0.61538 for $0.635 < r < 3.4925$ (see Figure B.3).
2. Concrete: the trace elements (S and K) were neglected. The following isotopes are not available and must be represented in the calculation in another fashion: H, Si, Ca, and Mg. This substitution will not significantly affect the calculation since this is not a shielding calculation. The concrete is merely serving the purpose of providing a lower boundary condition for the calculation. The following representation was made by increasing the number density of carbon to such an

extent as to create the same slowing down power ($\xi \Sigma_s$) in carbon as the simulated elements would have created. The following table illustrates this procedure:

Isotope	$\sim \xi$	$\sim \sigma_s$	N	$\xi \Sigma_s$	
H	1.0	17	.012048	.204816	} .20724
Si	.0698	2.3	.005924	9.5073×10^{-4}	
Ca	.0492	2.5	.008747	1.07593×10^{-3}	
Mg	.0811	3.5	.001413	4.01079×10^{-4}	
C	.158	4.7	.005924	4.39886×10^{-3}	

Additional C needed: .279079 (~ 23 times N^H)

Thus the concrete was mocked-up with the following number densities:

C-12: 2.85002×10^{-1} Na-23: 2.88104×10^{-4} Al-27: 1.89644×10^{-3}
 O-16: 4.20174×10^{-3} Fe: 2.80487×10^{-4}

3. Structural Steel (1020):

Fe: 8.437036×10^{-2} Mn-55: 3.821506×10^{-4}
 C-12 1.6984×10^{-4}

4. Inner Transformer Region:

Volume fraction fuel: 0.7352

Volume fraction clad: 0.1487

Fuel (94% theoretical density):

U-235: 1.1033×10^{-3} U-238: 2.18825×10^{-2} O-16: 4.59716×10^{-2}

Steel (SS-348) number densities are listed later.

5. Outer Transformer Region:

Volume Fraction	Material
.3246	Fuel (94% theoretical density)
.4684	Sodium filler in interstitial can
.0657	Clad (SS-348)
.0894	Interstitial can wall (SS-316)

• Fuel:

U-235: 1.103238×10^{-3} U-238: 2.188088×10^{-2} O-16: 4.59682×10^{-2}

Sodium:

Na-23: 2.54056×10^{-2}

Steel number densities listed later.

6. Blanket Region (Natural Uranium):

Volume Fraction	Material
.597	Fuel (94% theoretical density)
.115148	Glad (Aluminum)
.173	Steel Sleeves (SS-316)

Fuel:

U-235: 1.65479×10^{-4} U-238: 2.281778×10^{-2} O-16: 4.59665×10^{-2}

Glad:

Al-27: 6.02802×10^{-2}

7. Reflector Region:

66% Structural Steel (see material #3)

34% Na (Chlorine unavailable for the salt mock-up)

Sodium:

Na-23: 2.54056×10^{-2}

Steel Compositions:

SS-348
Fe: 6.06711×10^{-2} Ni: 9.53403×10^{-3} Cr: 1.56012×10^{-2} Mn-55: 8.66730×10^{-4}

SS-316
Fe: 5.69694×10^{-2} Cr: 1.43479×10^{-2} Mo: 2.10998×10^{-2} Ni: 1.01279×10^{-2}

Steel Compositions - Nominal Values, Trace Elements Ignored

Type/Wt. %							Density
	Fe	Cr	Ni	Mn	Mo	Mol. Wt.	g/cm ³
SS-348	70	18	11	1	-	55.46	7.98
SS-316	67.5	17	12	2.5	-	56.43	7.92

3. One Dimensional Calculation

The one-dimensional calculation was performed with LAZARUS¹. LAZARUS is a one-dimensional diffusion theory, perturbation code which is based upon QX1² (which in turn, is based upon the widely used fast reactor code, MACH1³). It contains a cross section collapsing routine which produces few-group macroscopic cross sections, thus it was chosen as the one-dimensional analysis code.

The 21 group isotopic cross section set was provided by Argonne National Laboratory (its group structure is listed in Table B.1). It was created by the MC²-2 by using the homogenized number densities of four regions (IC, OC, B, and R) of the CRBR plant. This data set was provided in ISOTX format (see Ref. 6) and processed into LAZARUS format by the X-PROC⁶ code. The individual isotopes which are available in this cross section set are listed in Table B.2.

A one-dimensional run was made at room temperature using the geometry and material compositions at the FBBF midplane. This is represented by a height of about 97 cm in Fig. B.2. The source region was split into two distinct regions: one representing the source pellets and steel can (outer radius of 1.905 cm), and one representing the outer steel can wall (outer radius of 3.4925 cm - see Fig. B.3). A ²⁵²Cf source of 1.0×10^{10} n/sec was placed at the source region center line (the Cf-252 spectrum is listed in Table B.4), and the one-dimensional flux solution was obtained. This one-dimensional regional spatial flux was used to collapse the regional macroscopic cross sections from 21 groups to 6 groups. The collapsing procedure is to weight the "fine" group cross sections with the spatial flux integral for that "fine" group, and then to sum over all "fine" groups in the particular "broad" group of interest. Thus the macroscopic cross section of type x in region R for broad group g is defined as:

TABLE B.1

21 Group Structure

Group	Lower Energy Bound	Lower Lethargy Bound	Δu
1	6.065	0.5	.5
2	3.679 MeV	1.0	.5
3	2.231	1.5	.5
4	1.353	2.0	.5
5	.821	2.5	.5
6	.498	3.0	.5
7	.302	3.5	.5
8	.183	4.0	.5
9	.111	4.5	.5
10	67.380 KeV	5.0	.5
11	40.868	5.5	.5
12	19.793	6.225	.725
13	9.119	7.0	.775
14	5.531	7.5	.5
15	3.355	8.0	.5
16	2.0347	8.5	.5
17	1.234	9.0	.5
18	.7485	9.5	.5
19	.4540	10.0	.5
20	275.36 eV	10.5	.5
21	0.0	---	---

10 MeV upper energy limit

TABLE B.2
ISOTOPE INFORMATION

<u>Nuclide Identifier</u>	<u>Available for indicated regions</u>
U-35	C,O,B
U-36	C,O,B
U-38	C,O,B
PU38	C,O,B
PU39	C,O,B
PU40	C,O,B
PU41	C,O,B
PU42	C,O,B
NA23	C,O,B,R
O-16	C,O,B
FE__	C,O,B,R
CR__	C,O,B,R
NI__	C,O,B,R
MN55	C,O,B,R
MO__	C,O,B,R
AL27	C,O,B,R
C-12	C,O,B
B-10	C,O,B
B-11	C,O,B
CS35	C,O,B
XE35	C,O,B
XE36	C,O,B
SM49	C,O,B
SM50	C,O,B
NP 37	C,O,B
AM41	C,O,B
AM43	C,O,B
CM44	C,O,B

Note - C = Inner Core, O = Outer Core, B = Blanket, R = Reflector,

$$\bar{\Sigma}_{x,g}^R = \frac{\sum_{j \in G} \Sigma_{x,j} \int_R \phi_j(r) r dr}{\sum_{j \in G} \int_R \phi_j(r) r dr} \quad (B.1)$$

The transport cross section is also collapsed in this fashion, from which the diffusion coefficient is formed. Since only the macroscopic quantities are being collapsed, this procedure for collapsing the transport cross section is adequate.

The collapsing procedure produces accurate broad group cross sections for the materials which are in the one-dimensional mock-up of the FBBF. Since the one-dimensional calculation produces the flux solution at the facility's midplane the collapsing procedure yields directly the data for only the five midplane regions (and thus these five materials: source steel, inner and outer transformer, blanket, and reflector). Cross sections for materials contained in axial regions of the two-dimensional representation (e.g., concrete, structural steel, etc.) of the FBBF have been collapsed in the following way: the material compositions of these materials were placed in appropriate midplane regions of the facility (scaled by 10^{-15} so as to leave the flux shape unchanged) and collapsed over the spatial flux solution for that region. A list of these regional placements is provided below in the two-dimensional description (section 4). It was deemed advantageous to provide an additional spatial region for collapsing purposes by dividing the blanket region into two regions (and thus providing 2 regions over which cross sections could be collapsed instead of only 1). This was done arbitrarily at a radius of 42 cm and is indicated by a dotted line in Figure B.2. All blanket regions, however, are identical in composition.

The two-dimensional calculation was performed with 2DB⁷, a multigroup diffusion theory code for fast reactor analysis. The 6-group structure used for the 2DB run is listed in Table B.3 and the 6-group Cf-252 source spectrum is listed in Table B.4. The macroscopic cross sections which were used in this calculation were collapsed in either their corresponding region of the one-dimensional calculation, or in a region which contained a representative spectrum. The following is a list of the 2DB materials which fall into this category and the region of the one-dimensional case in which they were collapsed (refer to Figure B.1):

<u>2DB Composition</u>	<u>Collapsed in the 1-D Region</u>
Source Plug	Source Pellets + Steel
Axial Blanket	"Inner" Blanket
(radial blanket composition)	
Structural Steel	"Inner" Blanket
Concrete	"Outer" Blanket
Source Steel	Source Pellets + Steel

The total source was 10^{10} n/sec; in this two-dimensional representation it was distributed volumetrically throughout the first mesh interval in the source region (outer radius of 1.1642 cm). The axial mesh intervals had the appropriate source density placed into them in order to produce a cosine shape in the axial direction and yet maintain a total source strength of 10^{10} n/sec.

Source-free calculations were done in order to verify the accuracy of the 6-group cross section set and to compare the one-dimensional run versus the two-dimensional calculations. The one-dimensional run with 2DB was obtained by removing the axial flux dependency with zero gradient boundary conditions at the top and bottom of the middle region, and by imposing an axial buckling value which was identical to the one employed by LAZARUS. These results are listed below:

TABLE B.3

2DB 6 Group Structure

<u>Group</u>	<u>Lower Energy Bound</u>	<u>Lower Lethargy Bound</u>	<u>Δu</u>	<u>21 Group Set Boundaries</u>
1	0.821 MeV	2.5	2.5	1-5
2	183. KeV	4.0	1.5	6-8
3	40.868 KeV	5.5	1.5	9-11
4	5.531 KeV	7.5	2.0	12-14
5	1.234 KeV	9.0	1.5	15-17
6	0.0	-	-	18-21

10 MeV upper energy limit

TABLE B.4

Cf-252 Source Spectrum

I. 21 Group Structure

1	.046142	8	.027696
2	.147862	9	.0134287
3	.22920	10	.006551
4	.21936	11	.0031554
5	.157416	12	.00150818
6	.095311	13	.00016272
7	.052207	14	-

II. 6 Group Structure

1	0.79998	3	0.0231351
2	0.175214	4	0.0016709

Code	Energy Group	Dimensions	k_{eff}	%change $\left(\frac{\delta k}{k}\right)$	32
LAZARUS	(21)	1	.3989	-	
2DB	(6)	1	.3941	-1.2	
2DB	(6)	2(r-z)	.3929	-1.5	

The difference in k_{eff} between the 1-D LAZARUS run and the 1-D 2DB run can be attributed to the different cross sections (21 gp. vs. 6 gp. collapse) and to the differences in the finite difference solution (mesh points in LAZARUS vs. mesh intervals in 2DB).

The final neutron balance table from the 2DB source calculations is reproduced here for completeness. It represents the totals for the entire facility (as represented in Figure B.2 and rotated about the center line). IL, RL, TL, and BL stand for Left Leakage, Right Leakage, Top Leakage, and Bottom Leakage; respectively. Also available are the 6-group and total fluxes, by mesh interval; and the power density (MWt/liter), also by mesh interval.

NEUTRON BALANCE TABLE

Reaction Rates

<u>Group</u>	<u>Fission Source</u>	<u>Fission Rate</u>	<u>Absorption</u>	<u>In-Scatter*</u>	<u>Out-Scatter</u>
1	6.7284×10^9	1.5548×10^9	1.9633×10^9	7.9998×10^9	1.2751×10^{10}
2	1.9189×10^9	4.1583×10^8	1.7511×10^9	1.3401×10^{10}	1.3511×10^{10}
3	2.7737×10^8	5.2240×10^8	3.4529×10^9	1.4709×10^{10}	1.1447×10^{10}
4	3.9813×10^7	5.2774×10^8	6.2445×10^9	1.1576×10^{10}	5.2442×10^9
5	1.6648×10^6	2.4285×10^8	2.6736×10^9	5.2670×10^9	2.5651×10^9
6	1.7657×10^5	2.4175×10^8	2.4709×10^9	2.5668×10^9	0.0
Total	8.9663×10^9	3.50537×10^9	1.8556×10^{10}	5.5520×10^{10}	4.5519×10^{10}

$$\frac{\text{Fission-source rate}}{\text{Fission rate}} = 2.558 = \bar{\nu}$$

*Note - In-Scatter term includes the external source.

Group	L.L.	R.L.	T.L.	B.L.	Total Leakage
1	0.	4.9699×10^6	8.5190×10^6	4.005×10^4	1.3530×10^7
2	0.	2.8556×10^7	2.8548×10^7	2.0490×10^5	5.730×10^7
3	0.	4.1631×10^7	4.4635×10^7	2.8007×10^5	8.6546×10^7
4	0.	7.8532×10^7	4.8376×10^7	4.8903×10^5	1.2740×10^8
5	0.	1.5405×10^7	1.4364×10^7	2.3215×10^5	3.0000×10^7
6	0.	3.2359×10^7	9.0330×10^6	5.4739×10^7	9.6132×10^7
Total	0.	2.0145×10^8	1.5347×10^8	5.5987×10^7	4.1091×10^8

Integral Neutron BalanceNeutron ProductionCf - source: 10^{10} n/secAdditional fission neutrons: 8.97×10^9 n/secTotal neutron production: 1.897×10^{10} n/sec

Comparison with results of source multiplication formula:

$$S_{\text{total}} = \frac{S}{1 - k_{\text{eff}}} = \frac{10^{10}}{0.6071} = 1.647 \times 10^{10} \text{ n/sec};$$

i.e. the actual total neutron source is about 15% larger than the estimate of the source multiplication formula due to the fact that the actual spectrum is much harder near the Cf source than it appears in the k_{eff} eigenvalue calculation.

Neutron Losses

	n/sec	% of S_{total}
Absorption:	1.856×10^{10}	97.835
Leakage		
a) top-leakage, group 1:	8.52×10^6	0.045%
b) other leakage	4.024×10^8	2.121%
		100.00 %

List of References TASK B

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Task C:

The objectives of this task are: (1) loading the first configuration of the FBBF; (2) testing experimental equipment and techniques on the FBBF; and (3) performing check-out tests required for licensing purposes and for safe operation of the FBBF. Progress on this task is summarized below.

C.1. First FBBF Loading (P. J. Fulford, F. M. Clikeman, and R. H. Johnson)

Pre-analysis for the first loading is being performed and is discussed under Task B. The first loading will begin after construction of the FBBF is completed.

C.2. Testing Experimental Equipment and Techniques (F. M. Clikeman, R. H. Johnson, K. R. Boldt, D. W. Vehar, and J. L. Westbrook)

Actual testing on the FBBF will not be possible until the first loading is begun. Preliminary testing of equipment has been carried out, however. One purpose of this preliminary testing has been to identify equipment needs (see Task D). Preliminary testing for neutron spectroscopy and foil activation measurements is underway. Preliminary testing of thermoluminescent dosimeters has been completed. (J. L. Westbrook, "An Investigation of the Use of Thermoluminescent Dosimeters to Measure Gamma Heating in the Fast Breeder Blanket Facility," M.S. Thesis, Purdue University, 1976.)

C.3. Safety and Licensing Tests (F. M. Clikeman, P. J. Fulford, and R. H. Johnson)

The safety of the FBBF is included as an objective of this task for both licensing and operational reasons. Calculations showing the safety of the first FBBF loading (fuel and neutron source) are required by the Nuclear Regulatory Commission for the initial license. More importantly, safety calculations are used to determine that future FBBF loadings (such as enriched fuel in the blanket region and a stronger neutron source) will not lead to operational limitations on the facility.

One concern has been the development of a method of opening fuel rods for the purpose of inserting foils and gamma ray dosimeters between the fuel pellets. A special shielded hood has been designed to protect personnel assembling and disassembling the fuel rods with the experimental foils and dosimeters. Protection is provided both from the loose UO_2 dust that might be present in the fuel rods and from the activity that will be present due to the irradiation of the fuel in the FBBF.

A major safety test (after construction) will be the measurement of radiation dose rates outside the FBBF shield. A major operational concern is the radiation dose rate inside the FBBF shield, both during operation and following shutdown. (Personnel will be entering the shield shortly after shutdown to remove experiments.) Calculated estimates of these radiation dose rates for the initial FBBF loading are a required part of the licensing procedure. These estimates have been completed and have been submitted to the Nuclear Regulatory Commission; these estimates are summarized below:

C.3.a. Dose Rates Outside Concrete Shield (R. H. Johnson and P. J. McDaniel)

Radiation dose rates in the FBBF laboratory (room B-28 of the Physics Building, Fig. C.1) and in nearby portions of the Physics Building were calculated as part of the evaluation of the radiation safety of the facility.

A thick concrete shield (the walls and ceiling of room B-28C, Fig. C.1) is involved, so the problem is one of deep penetration and radiation transport considerations are important. Additional shielding is present in the form of the concrete ceiling of room B-28.

A removal theory approach was used to determine the significant radiation transport properties of the shield. However, capture gamma-ray doses are significant for such a thick shield; therefore, additional modeling was required.

The shielding calculations were based on parameters derived from one dimensional transport calculations performed using the discrete ordinates code ANISN¹. Figure C.2 is presented here as extracted from Ref. 2.

Figure C.2 is a calculation of the shielding properties of 03 concrete for a ²⁵²Cf source in one dimension.

RADIATION TRANSPORT - The basic method of describing the radiation transport in the present shielding calculations is a point kernel approach. The fast neutron dose rate is treated as a simple kernel given by

$$D_{FN} = \frac{G_{FN} S_o e^{-\lambda_R x}}{4 \pi r^2} \quad (C.1)$$

λ_R = Macroscopic removal cross section.

S_o = Source strength in neutrons/sec.

G_{FN} = Fast neutron flux to dose rate conversion factor.

r = Distance from point source (cm).

x = Shield thickness. (cm).

Figure C.1 Side view of rooms B-28, B-28C, 114 and 114A of the Physics Building.

FIG. C.2 TYPE 03 CONCRETE - FAST NEUTRON, THERMAL NEUTRON, PRIMARY GAMMA, AND CAPTURE GAMMA DOSE RATE IN TYPE 03 CONCRETE FROM A POINT ISOTROPIC FISSION SOURCE OF ^{252}Cf

The dose rate curves in Fig. C.2 for fast neutrons, thermal neutrons, capture gammas, and primary gammas were fit to determine effective removal cross sections for a shield thickness of 80 cm. For fast neutrons, the effective removal cross section was found to be 0.0874 cm^{-1} . For comparison, the removal cross section for 03 concrete was calculated from the removal cross sections reported in Ref. 3 and the elemental number densities for 03 concrete. The calculated value was 0.0840 cm^{-1} . The value of 0.0874 cm^{-1} was used for the shielding calculations. The calculations shown in Fig. C.2 are for a fission source. The actual neutron leakage spectrum from the FBBF is much softer; with this difference taken into consideration, a fast neutron flux to dose rate conversion factor of

$$G_{\text{FN}} = 0.150 \frac{\text{mrem/hr}}{\text{n/cm}^2/\text{sec}}$$

was used.

The thermal neutron, capture gamma, and primary gamma dose rates shown in Fig. C.2 vary linearly with the fast neutron source strength. An effective removal cross section and an effective flux to dose rate conversion factor were therefore determined for each component of the radiation field. The dose rate due to thermal neutrons is modeled as

$$D_{\text{TN}} = \frac{0.064 S_0 e^{-0.0874x}}{4\pi r^2} \quad (\text{C.2})$$

The dose rate due to primary gammas is modeled as

$$D_{\text{PG}} = \frac{0.003 S_0 e^{-0.0782x}}{4\pi r^2} \quad (\text{C.3})$$

The fast neutron leakage from the FBBF used in the shielding calculations was calculated (see Task B) for neutron energies greater than 0.8 MeV.

However, intermediate energy neutron leakage was also considered. The only significant contribution of intermediate energy neutrons is to the capture gamma dose rate. This contribution was calculated using the approximation that these intermediate neutrons thermalize immediately upon entering the shield. The capture gamma dose rate therefore includes contributions from both fast and intermediate neutrons and is modeled as

$$D_{CG} = \frac{0.099 S_o e^{-0.0628x}}{4\pi x^2} \quad (C.4)$$

Equations (C.1), (C.2), (C.3), and (C.4) are valid only for thick shields (≥ 50 cm thick). These equations yield dose rates in units of mrem/hr. Using the calculated leakage from the FBBF dose rates at the outer surface of the shield can be easily calculated.

At points external to the shield, the radiation emerging from the shield must be considered as a source. For rooms 114 and 114A, (Figs. C1 and C3) this was done by using the top of the FBBF shield as a source and assuming a $\cos \theta$ dependence for the angular flux (θ is the angle from normal to the shield).

A direct dose rate contribution of radiation emerging from the shield is calculated applying the effective point kernel outlined above and integrating over the area of the top of the shield.

In much of rooms 114 and 114A, however, the direct radiation from the FBBF shield is negligible compared to the small amount of radiation scattered from the walls and floor of those rooms. Scattered radiation dose rates are calculated by treating the walls and floor as volume sources and integrating over the volume. Self-shielding is calculated for the floor but not for the walls. The integration was done numerically using a Gaussian quadrature rule.

Figure C.3: Sketch of floor plan of rooms 114 and 114A of Physics Building. The symbol indicates the center line of the FBBF. The dashed line indicates the location of the southeast wall of room B-28, the FBBF laboratory.

The preceding discussion has outlined the model used for the FBBF shielding calculations. The source strengths used for these calculations were obtained from calculations discussed in Task B.

RESULTS - The calculated dose rates are summarized in Table C.1. Note that these dose rates have been calculated for a ^{252}Cf source of 10^{10} neutrons/sec. The actual source strength will probably be within 20% of this value; this uncertainty is due to the difficulty of fabricating ^{252}Cf sources to greater accuracy, although the source strength can be measured to an accuracy of several percent. The dose rates, of course, vary linearly with the source strength.

The extremely low dose rates calculated in this study lead to the conclusion that the shielding provided for the FBBF is sufficient to ensure the radiation safety of the the FBBF.

C.3.b. Dose Rates Inside Concrete Shield (R. H. Johnson)

The dose rates within the FBBF concrete shield were calculated based on the two-dimensional diffusion calculation mentioned above. During operation with the source of 10^{10} neutrons/sec, the calculated dose rate averaged over the side surface of the FBBF is 110 mrem/hr due to neutron leakage. Immediately following shutdown of the FBBF after an extended period of operation, the calculated dose rates are 0.4 mrem/hr due to delayed neutrons, 0.09 mrem/hr due to ^{24}Na activity in the reflector, and 0.01 mrem/hr due to ^{38}Cl activity in the reflector. One minute after shutdown, the calculated delayed neutron dose rate has decreased to 0.01 mrem/hr; after one minute, the dose rate within the shield will be due mainly to the ^{24}Na activity (15 hour half-life) in the reflector.

TABLE C.1

<u>SIDE SHIELD POINT</u>	DOSE RATE FOR	
	10^{10}	NEUTRONS/SEC SOURCE
TOTAL DOSE		0.0111 mrem/hr
FAST NEUTRON		0.0018 mrem/hr
THERMAL NEUTRON		0.0007 mrem/hr
CAPTURE GAMMA		0.0086 mrem/hr
<u>TOP SHIELD POINT</u>		
TOTAL DOSE		0.0236 mrem/hr
FAST NEUTRON		0.0037 mrem/hr
THERMAL NEUTRON		0.0016 mrem/hr
CAPTURE GAMMA		0.0183 mrem/hr
<u>PREPARATION ROOM</u>		
MAXIMUM TOTAL DOSE		0.00061 mrem/hr
AVERAGE TOTAL DOSE		0.00017 mrem/hr
<u>LECTURE AREA OF LECTURE HALL</u>		
MAXIMUM TOTAL DOSE		0.00059 mrem/hr
AVERAGE TOTAL DOSE		0.00012 mrem/hr
<u>SEATING AREA OF LECTURE HALL</u>		
MAXIMUM TOTAL DOSE		0.0000130 mrem/hr
AVERAGE TOTAL DOSE		0.0000033 mrem/hr

Task C

List of References

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